analysis and recommendation on the proposed action—renewal of the operating licenses for IP2 and IP3. The FSEIS is available in ADAMS under package Accession No. ML103360205. On June 20, 2013, the NRC staff issued a supplement to the FSEIS, updating its final analysis to include corrections to impingement and entrainment data presented in the FSEIS, revised conclusions regarding thermal impacts based on newly available thermal plume studies, and an update of the status of the NRC's consultation under section 7 of the Endangered Species Act with the National Marine Fisheries Service regarding the shortnose sturgeon and Atlantic sturgeon. The supplement to the FSEIS is available in ADAMS under Accession No. ML13170A028.

The purpose of this document is to inform the public that the NRC will be preparing a second supplement to the FSEIS to provide information to decision makers relevant to environmental impacts of the proposed federal action and to further the purposes of NEPA, including new aquatic impact data, refined cost estimates associated with the licensee's SAMA analysis, and other matters.

Dated at Rockville, Maryland, this 26th day of August, 2014.

For the Nuclear Regulatory Commission.

Elaine M. Keegan,

Acting Chief, Projects Branch 2, Division of License Renewal, Office of Nuclear Reactor Regulation.

[FR Doc. 2014–20810 Filed 8–29–14; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[NRC–2014–0193]

Biweekly Notice; Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations

AGENCY: Nuclear Regulatory Commission.

ACTION: Biweekly notice.

SUMMARY: Pursuant to Section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (NRC) is publishing this regular biweekly notice. The Act requires the Commission to publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license or combined license, as applicable, upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 7, 2014 to August 20, 2014. The last biweekly notice was published on August 19, 2014.

DATES: Comments must be filed by October 2, 2014. A request for a hearing must be filed by November 3, 2014.

ADDRESSES: You may submit comments by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- Federal Rulemaking Web site: Go to *http://www.regulations.gov* and search for Docket ID NRC–2014–0193. Address questions about NRC dockets to Carol Gallagher; telephone: 301–287–3422; email: *Carol.Gallagher@nrc.gov.* For technical questions, contact the individual listed in the **FOR FURTHER INFORMATION CONTACT** section of this document.
- Mail comments to: Cindy Bladey, Office of Administration, Mail Stop: 3WFN–06–A44M, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

For additional direction on obtaining information and submitting comments, see ''Obtaining Information and Submitting Comments'' in the **SUPPLEMENTARY INFORMATION** section of this document.

FOR FURTHER INFORMATION CONTACT: Angela Baxter, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington DC 20555–0001; telephone: 301–415–2976, email: *Angela.Baxter@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID NRC–2014–0193 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- Federal Rulemaking Web site: Go to *http://www.regulations.gov* and search for Docket ID NRC–2014–0193.
- NRC's Agencywide Documents Access and Management System (ADAMS): You may obtain publicly-available documents online in the ADAMS Public Documents collection at *http://www.nrc.gov/reading-rm/adams.html.* To begin the search, select ''*ADAMS Public Documents''* and then select ''*Begin Web-based ADAMS Search.''* For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr.resource@nrc.gov.* The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in the **SUPPLEMENTARY INFORMATION** section.
- NRC's PDR: You may examine and purchase copies of public documents at the NRC's PDR, Room O1–F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments

Please include Docket ID NRC–2014–0193 in the subject line of your comment submission, in order to ensure that the NRC is able to make your comment submission available to the public in this docket.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC posts all comment submissions at *http://www.regulations.gov* as well as entering the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment submissions into ADAMS.

II. Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Combined Licenses and Proposed No Significant Hazards Consideration Determination

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in § 50.92 of Title 10 of the *Code of Federal Regulations* (10 CFR), this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

A. Opportunity To Request a Hearing and Petition for Leave To Intervene

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license or combined license. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Agency Rules of Practice and Procedure'' in 10 CFR Part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the NRC's PDR, located at One White Flint North, Room O1–F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The NRC's regulations are accessible electronically from the NRC Library on the NRC's Web site at *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: (1) The name, address, and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also identify the specific contentions which the requestor/petitioner seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the requestor/petitioner shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the requestor/petitioner intends to rely in proving the contention at the hearing. The requestor/petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the requestor/petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the requestor/petitioner to relief. A requestor/petitioner who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, then any hearing held would take place before the issuance of any amendment.

B. Electronic Submissions (E-Filing)

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC's E-Filing rule (72 FR 49139; August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least ten 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by email at *hearing.docket@nrc.gov,* or by telephone at 301–415–1677, to request (1) a digital identification (ID) certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/getting-started.html.* System requirements for accessing the E-Submittal server are detailed in the NRC's ''Guidance for Electronic Submission,'' which is available on the agency's public Web site at *http://www.nrc.gov/site-help/e-submittals.html.* Participants may attempt to use other software not listed on the Web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, Web-based submission form. In order to serve documents through the Electronic Information Exchange System, users will be required to install a Web browser plug-in from the NRC's Web site. Further information on the Web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals.html.*

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals.html.* A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an email notice confirming receipt of the document. The E-Filing system also distributes an email notice that provides access to the document to the NRC's Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically using the NRC's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the ''Contact Us'' link located on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals.html,* by email to *MSHD.Resource@nrc.gov,* or by a toll-free call at 1–866–672–7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket which is available to the public at *http://ehd1.nrc.gov/ehd/,* unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. However, a request to intervene will require including information on local residence in order to demonstrate a proximity assertion of interest in the proceeding. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

Petitions for leave to intervene must be filed no later than 60 days from the date of publication of this notice. Requests for hearing, petitions for leave to intervene, and motions for leave to file new or amended contentions that are filed after the 60-day deadline will not be entertained absent a determination by the presiding officer that the filing demonstrates good cause by satisfying the three factors in 10 CFR 2.309(c)(1)(i)–(iii).

For further details with respect to this license amendment application, see the application for amendment which is available for public inspection in ADAMS and at the NRC's PDR. For additional direction on accessing information related to this document, see the ''Obtaining Information and Submitting Comments'' section of this document.

Duke Energy Progress Inc., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, New Hill, North Carolina

Date of amendment request: June 19, 2014. A publicly-available version is in ADAMS under Accession No. ML14174A118.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3.2, ''Engineered Safety Features Actuation System Instrumentation,'' Table 3.3–4, ''Engineered Safety Features Actuation System Instrumentation Trip Setpoints.'' Specifically, the instrument trip setpoint and associated allowable value are being revised to ensure that the trip of the safety-related alternating current bus will occur at a voltage at or above the minimum voltage necessary to operate the applicable safety-related loads.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change revises the TS Table 3.3–4 Functional Unit 9.a, Loss-of-Offsite Power 6.9 kV Emergency Bus Undervoltage—Primary, instrumentation trip setpoint and allowable value. The Loss-of-Offsite Power, 6.9 kV Emergency Bus Undervoltage—Primary instrumentation is not an initiator to any accident previously evaluated. As such, the probability of an accident previously evaluated is not increased. The Loss-of-Offsite Power, 6.9 kV Emergency Bus Undervoltage—Primary instrumentation revised values continue to provide reasonable assurance that the Functional Unit 9.a will continue to perform its intended safety functions. As a result, the proposed change will not increase the consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change revises the TS Table 3.3–4 Functional Unit 9.a, Loss-of-Offsite Power 6.9 kV Emergency Bus Undervoltage—Primary, instrumentation trip setpoint and allowable value. No new operational conditions beyond those currently allowed are introduced. This change is consistent with the safety analyses assumptions and current plant operating practices. This simply corrects the setpoint consistent with the accident analyses and therefore cannot create the possibility of a new or different kind of accident from any previously evaluated accident.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change revises the TS Table 3.3–4 Functional Unit 9.a, Loss-of-Offsite Power 6.9 kV Emergency Bus Undervoltage—Primary, instrumentation trip setpoint and allowable value. Function 9.a protects the emergency power system against loss of voltage. This change is consistent with the safety analyses assumptions and current plant operating practices. No new operational conditions beyond those currently allowed are created by these changes.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Lara S. Nichols, Deputy General Counsel, Duke Energy Corporation, 550 South Tyron Street, Mail Code DEC45A, Charlotte, NC 28202.

NRC Acting Branch Chief: Lisa M. Regner.

Entergy Nuclear Operations, Inc., Docket No. 50–255, Palisades Nuclear Plant, Van Buren County, Michigan

Date of amendment request: June 11, 2014. A publicly-available version is in ADAMS under Accession No. ML14162A079.

Description of amendment request: The proposed amendment would modify technical specification (TS) requirements to adopt the changes described in TS Task Force (TSTF)-426, Revision 5, ''Revise or Add Actions to Preclude Entry into LCO [limiting condition for operation] 3.0.3—RITSTF [Risk-Informed TSTF] Initiatives 6b & 6c'' (ADAMS Accession No. ML113260461).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change provides a short Completion Time to restore an inoperable system for conditions under which the existing Technical Specifications require a plant shutdown to begin within 1 hour in accordance with LCO 3.0.3. Entering into Technical Specification Actions is not an initiator of any accident previously evaluated. As a result, the probability of an accident previously evaluated is not significantly increased. The consequences of any accident previously evaluated that may occur during the proposed Completion Times are no different from the consequences of the same accident during the existing 1 hour allowance. As a result, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No.

No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change increases the time the plant may operate without the ability to perform an assumed safety function. The analysis in WCAP–16125–NP–A, ''Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,'' Revision 2, August 2010, demonstrated that there is an acceptably small increase in risk due to a limited period of continued operation in these conditions and that the risk is balanced by avoiding the risks associated with a plant shutdown. As a result, the change to the margin of safety provided by requiring a plant shutdown within 1 hour is not significant.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Mr. William Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Ave., White Plains, NY 10601.

NRC Branch Chief: David L. Pelton.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

Date of amendment request: July 11, 2014. A publicly-available version is in ADAMS under Accession No. ML14192B143.

Description of amendment request: The proposed amendment would incorporate several miscellaneous administrative changes to the Facility Operating License and the Technical Specifications. For example, the amendment would delete historical items that are no longer applicable, correct errors, and remove references that are no longer valid.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

No physical changes to the facility will occur as a result of this proposed amendment. The proposed changes will not alter the physical design or operational procedures associated with any plant structure, system, or component. The proposed changes are administrative in nature and have no effect on plant operation.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes are administrative in nature. The proposed changes do not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant. Accordingly, the changes do not introduce any new accident initiators, nor do they reduce or adversely affect the capabilities of any plant structure, system, or component to perform their safety function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes conform to NRC regulatory guidance regarding the content of plant Technical Specifications. The proposed changes are administrative in nature. The proposed changes do not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for Licensee: J. Bradley Fewell, Vice President and Deputy General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

Acting NRC Branch Chief: Robert G. Schaaf.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

Date of application for amendments: July 10, 2014. A publicly-available version is in ADAMS under Accession No. ML14191B190.

Description of amendment request: The proposed amendment would revise and add Technical Specification (TS) surveillance requirements to address the concerns discussed in NRC Generic Letter 2008–01, ''Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems,'' dated January 11, 2008 (ADAMS Accession No. ML072910759). The proposed TS changes are based on NRC-approved TS Task Force (TSTF) Traveler TSTF–523, Revision 2, ''Generic Letter 2008–01, Managing Gas Accumulation,'' dated February 21, 2013 (ADAMS Accession No. ML13053A075). The NRC staff issued a Notice of Availability for TSTF–523, Revision 2, for plant-specific adoption using the Consolidated Line Item Improvement Process, in the **Federal Register** on January 15, 2014 (79 FR 2700).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change revises or adds Surveillance Requirements (SRs) that require verification that the Emergency Core Cooling Systems, the Suppression Pool Cooling System, the Suppression Pool Spray System, the Drywell Spray System, the Shutdown Cooling System, and the Reactor Core Isolation Cooling System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. Gas accumulation in the subject systems is not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The proposed SRs ensure that the subject systems continue to be capable of performing their assumed safety function and are not rendered inoperable due to gas accumulation. Thus, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change revises or adds SRs that require verification that the Emergency Core Cooling Systems, the Suppression Pool Cooling System, the Suppression Pool Spray System, the Drywell Spray System, the Shutdown Cooling System, and the Reactor Core Isolation Cooling System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the proposed change does not impose any new or different requirements that could initiate an accident. The proposed change does not alter assumptions made in the safety analysis and is consistent with the safety analysis assumptions.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change revises or adds SRs that require verification that the Emergency Core Cooling Systems, the Suppression Pool Cooling System, the Suppression Pool Spray System, the Drywell Spray System, the Shutdown Cooling System, and the Reactor Core Isolation Cooling System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. The proposed change adds new requirements to manage gas accumulation in order to ensure the subject systems are capable of performing their assumed safety functions. The proposed SRs are more comprehensive than the current SRs and will ensure that the assumptions of the safety analysis are protected. The proposed change does not adversely affect any current plant safety margins or the reliability of the equipment assumed in the safety analysis. Therefore, there are no changes being made to any safety analysis assumptions, safety limits or limiting safety system settings that would adversely affect plant safety as a result of the proposed change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for Licensee: J. Bradley Fewell, Esquire, Vice President and Deputy General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

Acting NRC Branch Chief: Robert G. Schaaf.

Exelon Generation Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of amendment request: July 10, 2014. A publicly-available version is in ADAMS under Accession No. ML14191A059.

Description of amendment request: The proposed amendment would revise and add Technical Specification (TS) Surveillance Requirements to address the concerns discussed in NRC Generic Letter 2008–01, ''Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems,'' dated January 11, 2008 (ADAMS Accession No. ML072910759). The proposed TS changes are based on NRC-approved TS Task Force (TSTF) Traveler TSTF–523, Revision 2, ''Generic Letter 2008–01, Managing Gas Accumulation,'' dated February 21, 2013 (ADAMS Accession

No. ML13053A075). The NRC staff issued a Notice of Availability for TSTF–523, Revision 2, for plant-specific adoption using the Consolidated Line Item Improvement Process, in the **Federal Register** on January 15, 2014 (79 FR 2700).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change adds Surveillance Requirements (SRs) that require verification that the Emergency Core Cooling System (ECCS), the Decay Heat Removal (DHR) System, and the Reactor Building Spray (RB Spray) System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. Gas accumulation in the subject systems is not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The proposed SRs ensure that the subject systems continue to be capable of performing their assumed safety function and are not rendered inoperable due to gas accumulation. Thus, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change adds SRs that require verification that the ECCS, the DHR, and the RB Spray System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the proposed change does not impose any new or different requirements that could initiate an accident. The proposed change does not alter assumptions made in the safety analysis and is consistent with the safety analysis assumptions.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change adds SRs that require verification that the ECCS, the DHR, and the RB Spray System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. The proposed change adds new requirements to manage gas accumulation in order to ensure that the subject systems are capable of performing their assumed safety functions. The proposed SRs are more comprehensive than the current SRs and will ensure that the assumptions of the safety analysis are protected. The proposed change does not adversely affect any current plant safety margins or the reliability of the equipment assumed in the safety analysis. Therefore, there are no changes being made to any safety analysis assumptions, safety limits, or limiting safety system settings that would adversely affect plant safety as a result of the proposed change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J. Bradley Fewell, Vice President and Deputy General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

Acting NRC Branch Chief: Robert G. Schaaf.

Exelon Generation Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: July 10, 2014. A publicly-available version is in ADAMS under Accession No. ML14191B180.

Description of amendment request: The proposed amendment would revise and add Technical Specification (TS) surveillance requirements to address the concerns discussed in NRC Generic Letter 2008–01, ''Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems,'' dated January 11, 2008 (ADAMS Accession No. ML072910759). The proposed TS changes are based on NRC-approved TS Task Force (TSTF) Traveler TSTF–523, Revision 2, ''Generic Letter 2008–01, Managing Gas Accumulation,'' dated February 21, 2013 (ADAMS Accession No. ML13053A075). The NRC staff issued a Notice of Availability for TSTF–523, Revision 2, for plant-specific adoption using the Consolidated Line Item Improvement Process, in the **Federal Register** on January 15, 2014 (79 FR 2700).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change revises or adds Surveillance Requirements (SRs) that require verification that the Emergency Core Cooling System (ECCS), the Residual Heat Removal (RHR) System, the Shutdown Cooling (SDC) System, the Containment Spray (CS) System, and the Reactor Core Isolation Cooling (RCIC) System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. Gas accumulation in the subject systems is not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The proposed SRs ensure that the subject systems continue to be capable of performing their assumed safety function and are not rendered inoperable due to gas accumulation. Thus, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change revises or adds SRs that require verification that the ECCS, the RHR, the SDC, the CS, and the RCIC Systems are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the proposed change does not impose any new or different requirements that could initiate an accident. The proposed change does not alter assumptions made in the safety analysis and is consistent with the safety analysis assumptions.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change revises or adds SRs that require verification that the ECCS, the RHR, the SDC, the CS, and the RCIC Systems are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. The proposed change revises or adds new requirements to manage gas accumulation in order to ensure the subject systems are capable of performing their assumed safety functions. The proposed SRs are more comprehensive than the current SRs and will ensure that the assumptions of the safety analysis are protected. The proposed change does not adversely affect any current plant safety margins or the reliability of the equipment assumed in the safety analysis. Therefore, there are no changes being made to any safety analysis assumptions, safety limits or limiting safety system settings that would adversely affect plant safety as a result of the proposed change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J. Bradley Fewell, Vice President and Deputy General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

Acting NRC Branch Chief: Robert G. Schaaf.

Exelon Generation Company, LLC, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Units 1 and 2, Will County, Illinois

Docket Nos. STN 50–454 and STN 50–455, Byron Station, Units 1 and 2, Ogle County, Illinois

Date of amendment request: April 24, 2014. A publicly-available version is in ADAMS under Accession No. ML14120A039.

Description of amendment request: The proposed amendment would add new ''low degraded voltage relays'' and timers, with appropriate settings, on each engineered safety feature electrical bus. The technical specifications and surveillance requirements would be changed to add appropriate operational and testing requirements for the new relays and timers.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, with NRC staff revisions provided in [brackets], which is presented below:

EGC [Exelon Generation Company, LLC] has evaluated the proposed change for Braidwood Station and Byron Station, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

Criteria

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change to add new ''low degraded voltage relays'' (LDVRs) and associated CHANNEL CALIBRATION surveillance test provides a third level of undervoltage protection for the Engineered Safeguards Features (ESF) electrical buses. These new relays will further ensure that the normally operating safety-related motors/equipment, which are powered from the ESF buses, are appropriately isolated from the normal off-site power source and will not be damaged in the event of sustained degraded bus voltage. The addition of the LDVRs will continue to allow the existing undervoltage protection circuitry to function as originally designed; i.e., the first-level ''loss of voltage'' protection and the second-level ''degraded voltage'' protection will remain in place and be unaffected by this change. The proposed change does not affect the probability of any accident resulting in a loss of voltage or degraded voltage condition on the ESF electrical buses; and will positively impact the consequences of accidents previously evaluated as this change further ensures continued operation of safety-related equipment throughout the accident scenarios.

Specific analysis was performed and determined that the proposed LDVRs, with the specified allowable values and time delay, will ensure that the 4.16 kV ESF buses will be isolated from the normal off-site power source, at the appropriate voltage level, under nonaccident sustained degraded voltage conditions. The normally operating safety related motors will be subsequently sequenced back on to the 4.16 kV ESF buses powered by the EDGs [Emergency Diesel Generators]; and therefore, will not be damaged in the event of sustained degraded bus voltage during the time delay period prior to initiation of the first level loss of voltage trip function.

Therefore, these safety-related loads will be available to perform their design basis function should a loss-of-coolant accident (LOCA) occur concurrent with a loss-of-offsite power (LOOP) following the degraded voltage condition. The loading sequence (i.e., timing) of safety-related equipment back onto the ESF bus, powered by the EDG, is not affected by the addition of the new LDVRs.

The addition of new LDVRs will have no impact on accident initiators or precursors; does not alter the accident analysis assumptions or the manner in which the plant is operated or maintained; and does not affect the probability of operator error.

Based on the above, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change involves the addition of new ''low degraded voltage relays'' (LDVRs); i.e., a third level of undervoltage protection for the ESF electrical buses, and adds an associated CHANNEL CALIBRATION surveillance test. This change helps ensure that the assumptions in the previously evaluated accidents, which may involve a degraded voltage condition, continue to be valid.

The proposed changes do not result in the creation of any new accident precursors; do not result in changes to any existing accident scenarios; and do not introduce any operational changes or mechanisms that would create the possibility of a new or different kind of accident. A specific failure mode and effects review was completed for the new LDVRs, considering their potential failure, and concluded that the addition of these relays would not affect the existing ''loss of voltage'' and ''degraded voltage'' protection schemes; would not affect the number of occurrences of degraded voltage conditions that would cause the actuation of the existing Loss of Voltage Relays (LVRs), Degraded Voltage Relays (DVRs) or new LVDRs; would not affect the failure rate of the existing protection relays; and would not impact the assumptions in any existing accident scenario.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The current ''loss of voltage'' and ''degraded voltage'' protection circuitry is designed to appropriately isolate the normally operating safety-related motors/equipment, which are powered from the ESF buses, from the normal off-site power source such that the subject equipment will not be damaged in the event of sustained degraded bus voltage. The loss of voltage relays (LVRs) isolate the ESF buses at a TS [technical specifications] voltage value of approximately 66% of the nominal bus value after a short time delay (i.e., 1.9 seconds); while the degraded voltage relays (DVRs) isolate the ESF buses at a TS voltage value of 94.5% for Braidwood (91.2% for Byron Station) of the nominal bus voltage after a longer time delay of up to 5 minutes and 40 seconds (if no safety injection signal is present). After the ESF buses are isolated from the offsite power supply, the normally operating safety related motors will be sequenced back on to the 4.16 kV EFS bus powered by the EDG; and continue to perform their design basis function to mitigate the consequences of an accident, with a specified margin of safety.

A concern exists that ESF motors/equipment may be damaged when operating and/or starting safety-related equipment when bus voltage drops to just above the loss of voltage relay setpoint for the duration of the 5 minutes and 40 second time delay. The new LDVRs are being added to resolve this concern. Analysis has been performed that shows the ESF equipment will not be damaged at 75% of bus voltage; therefore, the LDVR setpoint will be set at 75% of nominal ESF bus voltage. With the addition of this new third level of undervoltage protection, the capability of the ESF equipment will be assured; and thus the equipment will continue to perform its design basis function to mitigate the consequences of the previously analyzed accidents; and maintain the existing margin to safety currently assumed in the accident analyses.

An EDG start due to a safety injection signal (i.e., Loss of Coolant Accident) and the subsequent sequencing of ESF loads back on to the ESF buses, powered by the EDG, is not adversely affected by this change. If an actual loss of voltage condition occurs on the ESF buses, the loss of voltage time delays will continue to isolate the 4.16 kV ESF distribution system from the offsite power source prior to the EDG assuming the ESF loads.

The ESF loads will sequence back on to the bus in a specified order and time interval; again ensuring that the existing accident analysis assumptions remain valid and the existing margin to safety is unaffected.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, EGC concludes that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Attorney for licensee: Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Travis L. Tate.

NextEra Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit 1, Rockingham County, New Hampshire

Date of amendment request: June 24, 2014. A publicly-available version is in ADAMS under Accession No. ML14177A503.

Description of amendment request: The proposed amendment would revise and add Technical Specification (TS) Surveillance Requirements (SRs) to address the concerns discussed in NRC Generic Letter 2008–01, ''Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems,'' dated January 11, 2008 (ADAMS Accession No. ML072910759). The proposed TS changes are based on NRC-approved TS Task Force (TSTF) Traveler TSTF–523, Revision 2, ''Generic Letter 2008–01, Managing Gas Accumulation,'' dated February 21, 2013 (ADAMS Accession No. ML13053A075). The NRC staff issued a Notice of Availability for TSTF–523, Revision 2, for plant-specific adoption using the Consolidated Line Item Improvement Process, in the **Federal Register** on January 15, 2014 (79 FR 2700).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, with NRC staff revisions provided in [brackets], which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change revises or adds SRs that require verification that the Emergency Core Cooling Systems (ECCS), Residual Heat Removal (RHR) System, and Containment Spray (CS) System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. Gas accumulation in the subject systems is not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The proposed SRs ensure that the subject systems continue to be capable to perform their assumed safety function and are not rendered inoperable due to gas accumulation. Thus, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change revises or adds SRs that require verification that the ECCS, RHR System, and CS System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the proposed change does not impose any new or different requirements that could initiate an accident. The proposed change does not alter assumptions made in the safety analysis and is consistent with the safety analysis assumptions[.]

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change revises or adds SRs that require verification that the ECCS, RHR System, and CS System are not rendered inoperable due to accumulated gas and to provide allowances which permit performance of the revised verification. The proposed change adds new requirements to manage gas accumulation in order to ensure that the subject systems are capable of performing their assumed safety functions. The proposed SRs are more comprehensive than the current SRs and will ensure that the assumptions of the safety analysis are protected. The proposed change does not adversely affect any current plant safety margins or the reliability of the equipment assumed in the safety analysis. Therefore, there are no changes being made to any safety analysis assumptions, safety limits, or limiting safety system settings that would adversely affect plant safety as a result of the proposed change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: James Petro, Managing Attorney, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

Acting NRC Branch Chief: Robert G. Schaaf.

South Carolina Electric and Gas Docket Nos.: 52–027 and 52–028, Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3, Fairfield County, South Carolina

Date of amendment request: May 20, 2014, as supplemented by letter dated June 3, 2014. Publicly-available versions are in ADAMS under Accession Nos. ML14140A637 and ML14155A257, respectively.

Description of amendment request: The proposed change would amend Combined License Nos. NPF–93 and NPF–94 for VCSNS Units 2 and 3 by departing from the plant-specific Design Control Document (DCD) Tier 1(and corresponding Combined License Appendix C information) material by making various nontechnical changes to correct editorial and consistency errors in Tier 1. This is being done to promote consistency within the Updated Final Safety Analysis Report (UFSAR).

Because, this proposed change requires a departure from Tier 1 information in the Westinghouse Advanced Passive 1000 DCD, the licensee also requested an exemption from the requirements of the Generic DCD Tier 1 in accordance with 10 CFR 52.63(b)(1).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, with NRC staff revisions provided in [brackets], which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed editorial and consistency plant-specific Tier 1 and corresponding COL [combined operating license] Appendix C update does not involve a technical change, e.g., there is no design parameter or requirement, calculation, analysis, function or qualification change. No structure, system, or component (SSC) design or function would be affected. No design or safety analysis would be affected. The proposed changes do not affect any accident initiating event or component failure, thus the probabilities of the accidents previously evaluated are not affected. No function used to mitigate a radioactive material release and no radioactive material release source term is involved, thus the radiological releases in the accident analyses are not affected.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed editorial and consistency plant-specific Tier 1 and corresponding COL Appendix C update would not affect the design or function of any SSC, but will instead provide consistency between the SSC designs and functions currently presented in the UFSAR and the Tier 1 information. The proposed changes would not introduce a new failure mode, fault or sequence of events that could result in a radioactive material release.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed editorial and consistency plant-specific Tier 1 and corresponding COL Appendix C update is considered non-technical for reasons discussed above, thus would not affect any design parameter, function or analysis. There would be no change to an existing design basis, design function, regulatory criterion, or analysis. No safety analysis or design basis acceptance limit/criterion is involved.

Therefore, the proposed amendment does not reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Kathryn M. Sutton, Morgan, Lewis & Bockius LLC, 1111 Pennsylvania Avenue NW., Washington, DC 20004–2514.

NRC Branch Chief: Lawrence J. Burkhart.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2 (VEGP), Burke County, Georgia

Date of amendment request: August 31, 2012, as supplemented September 13, 2013, May 2, July 22, and August 11, 2014. Publicly-available versions are in ADAMS under Accession Nos. ML12248A035, ML13256A306, ML14122A364, ML14203A252 and, ML14223A616, respectively.

Description of amendment request: The proposed amendments would revise the licensing basis for the VEGP by adding license conditions that would allow for the voluntary implementation of 10 CFR 50.69, ''Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors.'' As indicated in § 50.69, a licensee may voluntarily comply with § 50.69 as an alternative to compliance with the following requirements for certain SSCs: (i) 10 CFR part 21, (ii) a portion of § 50.46, (iii) § 50.49, (v) certain requirements of § 50.55a, (vi) § 50.65, (vii) § 50. 72, (viii) § 50.73,·(ix) Appendix B to Part 50, (x) certain containment leakage testing requirements, and (xi) certain requirements of Appendix A to part 100.

Basis for proposed no significant hazards consideration determination: The licensee responded in its letter dated August 11, 2014, to the NRC staff's request for additional information regarding the licensee's no significant hazards consideration determination, which is required by 10 CFR 50.91(a). Portions of the licensee's response regarding each of the no significant hazards consideration standards, with NRC staff revisions provided in [brackets], are presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

Operation of the Vogtle Electric Generating Plant (VEGP) in accordance with the proposed amendment does not result in a significant increase in the probability or consequences of accidents previously evaluated. The Updated Final Safety Analysis Report (UFSAR) documents the analysis of design basis accidents at VEGP. The proposed amendment does not affect accident initiators, nor does it alter design assumptions, conditions, or configurations of the facility that would increase the probability of accidents previously evaluated, nor does it adversely alter design assumptions, conditions, or configurations of the facility, and it does not adversely impact the ability of structures, systems, or components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits, nor do they affect assumed failure modes for accidents described and evaluated in the UFSAR. The proposed changes do not affect the way in which required systems perform their functions as required by the accident analysis. Structures, systems, and components required to safely shut down the reactor and maintain it in a safe shutdown condition will remain capable of performing their design functions.

Furthermore, the source term and radiological release assumptions of previously evaluated events are not affected by the alternative treatments permitted under 10 CFR 50.69; containment isolation devices assumed to function under accident conditions will not have their reliability adversely affected by the proposed amendment. Consequently, operating under the proposed amendment will not result in a significant increase in the radiological dose consequences assumed for previously analyzed events.

Section 50.69 defines the terminology ''safety significant function'' as functions whose loss or degradation could have a significant adverse effect on defense-in-depth, safety margins, or risk. For SSCs determined to be safety significant, 50.69 maintains the current regulatory requirements. These current requirements are adequate for addressing design basis performance of these SSCs.

The purpose of this amendment is to permit VEGP to adopt a new risk-informed licensing basis for categorization and treatment of structures, systems and components. The proposed VEGP Units 1 and 2 OL [operating license] LCs [license conditions] will allow for the voluntary implementation of 10 CFR 50.69. The SNC [Southern Nuclear Operating Company] risk-informed categorization process has been documented per the requirements of 10 CFR 50.69(b)(2) and meets the requirements of 10 CFR 50.69(c). A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety and providing a logical means for prioritizing these challenges based on safety significance. The SNC risk-informed categorization process will be used to modify the scope of SSCs subject to special treatment requirements. Alternative treatments permitted per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. The process provides reasonable confidence that, for SSCs categorized as RISC–3, sufficient safety margins are maintained and that any potential increases in CDF [core damage frequency] and LERF [large early release frequency] resulting from changes in treatment are small per 10 CFR 50.69(c)(1)(iv). The proposed OL LCs do not result in or require any physical or operational changes to VEGP SSCs, including SSCs intended for the prevention or mitigation of accidents. Implementation of 10 CFR 50.69 in compliance with 10 CFR 50.69 requirements ensures that RISC–1 and RISC–3 SSCs remain capable of performing their design basis functions, including safety-related functions, under design basis conditions. In addition, the process ensures that RISC–2 SSCs are capable of performing their safety significant functions.

Based on the above, implementation of this amendment to implement 10 CFR 50.69 risk informed categorization and treatment of structures, systems, and components does not involve a significant increase in the probability of any accident previously evaluated. In addition, all equipment required to mitigate an accident remains capable of performing the assumed function.

Therefore, consequences of any accident previously evaluated are not significantly increased with the implementation of this License Amendment.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Operation of VEGP in accordance with the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment does not impact any scenario or previously analyzed accident with offsite dose consequences included in the evaluation of design basis accidents (DBA) documented in the FSAR [final safety analysis report]. The proposed change does not alter the requirements or functions for systems required during accident conditions, nor does it alter the required mitigation systems as assumed in the licensing basis analyses and/or DBA radiological consequences evaluations. Implementation of the 50.69 categorization will not result in new or different accidents.

The proposed amendment does not adversely affect accident initiators nor alter design assumptions, or conditions of the facility. The proposed amendment does not introduce new or different accident initiators; neither does it introduce new modes of operation. The proposed amendment does not adversely affect the ability of SSCs to perform their design function. SSCs required to safely shutdown the reactor and maintain it in a safe shutdown condition remain capable of performing their design function.

Section 50.69 represents an alternative set of requirements whereby a licensee may voluntarily undertake categorization of its SSCs consistent with the requirements in 50.69(c), remove the special treatment requirements listed in 50.69(b) for SSCs that are determined to be of low safety significance, and implement alternative treatment requirements in 50.69(d). The regulatory requirements not removed continue to apply. These requirements are adequate for addressing design basis performance of these SSCs. This license amendment continues to maintain the principles that the net increase in plant risk is small, defense-in-depth is maintained, and safety margins are maintained.

The proposed VEGP Units 1 and 2 OL LCs will allow for the voluntary implementation of 10 CFR 50.69. The SNC risk-informed categorization process has been documented per the requirements of 10 CFR 50.69(b)(2) and meets the requirements of 10 CFR 50.69(c). The SNC risk-informed categorization process will be used to modify the scope of SSCs subject to special treatment requirements. Alternative treatments permitted per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. The process provides reasonable confidence that, for SSCs categorized as RISC–3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small per 10 CFR 50.69(c)(1)(iv). The proposed OL LCs do not result in or require any physical or operational changes to VEGP SSCs, including SSCs intended for the prevention or mitigation of accidents. Implementation of 10 CFR 50.69 in compliance with 10 CFR 50.69 requirements ensures that RISC–1 and RISC–3 SSCs remain capable of performing their design basis functions, including safety-related functions, under design basis conditions. In addition, the process ensures that RISC–2 SSCs are capable of performing their safety significant functions. Therefore, even though there was not an individual evaluation done of every UFSAR accident with potential off-site dose consequences, it can be concluded that the SSCs, assumed to mitigate the consequences of any and all previously evaluated events, will not be adversely affected by the alternative treatments allowed under 10 CFR 50.69. Consequently, the dose consequences of previously analyzed events will not significantly increase as a result of the alternative treatment of SSCs. Additionally, implementation of 10 CFR 50.69 will not create new failure mechanisms that initiate new accidents because the process does not result in or require any physical or operational changes for VEGP SSCs nor does it alter the functions or functional requirements of those SSCs.

Based on this, implementation of the proposed amendment would not create the possibility of a new or different kind of accident from any kind of accident previously evaluated. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this amendment. There will be no adverse effect or challenges imposed on required systems as a result of this amendment.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Operation of VEGP in accordance with the proposed amendment does not involve a significant reduction in the margin of safety. Implementation of a new risk informed categorization and treatment of structures, systems, and components licensing basis that complies with the requirements of 10 CFR 50.69 does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by this change. The proposed amendment does not adversely affect existing plant safety margins or the reliability of equipment assumed in the UFSAR to mitigate accidents. The proposed change does not adversely affect the ability of SSCs to perform their design function. The 10 CFR 50.69 process provides reasonable confidence that SSCs categorized as RISC–1, RISC–2, and RISC–3 maintain sufficient safety margins. The proposed amendment does not adversely impact systems required to safely shutdown the plant and maintain it in a safe condition.

The proposed VEGP Units 1 and 2 OL LCs will allow for the voluntary implementation of 10 CFR 50.69. The SNC risk-informed categorization process has been documented per the requirements of 10 CFR 50.69(b)(2) and meets the requirements of 10 CFR 50.69(c). The SNC risk-informed categorization process will be used to modify the scope of SSCs subject to special treatment requirements. Alternative treatments permitted per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. Although there were no calculations or evaluations performed for the express purpose of demonstrating that the implementation of 10 CFR 50.69 will not result in a significant reduction in the margin of safety, the process provides reasonable confidence that, for SSCs categorized as RISC–3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small per 10 CFR 50.69(c)(1)(iv). The only requirements that are relaxed for SSCs, consistent with their categorization, are those related to treatment. The safety margins associated with SSCs design basis functions and design technical requirements remain unchanged. Additionally, it is required that there be reasonable confidence that any potential increases in CDF and LERF be small from assumed changes in reliability resulting from the treatment changes permitted by 10 CFR 50.69. As a result individual SSCs continue to be capable of performing their design basis functions. It is concluded that sufficient safety margins are preserved.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Leigh D. Perry, SVP & General Counsel, Southern Nuclear Operating Company, 40 Inverness Center Parkway, Birmingham, AL 35242.

NRC Branch Chief: Robert Pascarelli.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2 (HNP), Appling County, Georgia

Date of amendment request: August 15, 2014. A publicly-available version is in ADAMS under Accession No. ML14227A921.

Description of amendment request: The proposed amendments would modify Technical Specification (TS) 3.8.7 to add two new safety-related instrument buses to the HNP electrical distribution system. Certain instruments will be re-located from existing safety-related electrical instrument buses to these new ''critical instrumentation buses.'' The existing instrument bus is listed in TS 3.8.7 of the HNP, Units 1 and 2, TSs and, since some of the instruments powered from this bus will be moved to the critical instrumentation bus, the new bus will be added to the list of the existing electrical buses in TS 3.8.7.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided an analysis of the issue of no significant hazards consideration, with NRC staff revisions provided in [brackets], as presented below:

Southern Nuclear Operating Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, ''Issuance of Amendment,'' as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously identified?

Response: No.

These new critical instrumentation buses and their inverters are not intended for the prevention of any previously analyzed transient or accident. They are intended to provide power to instruments which may be necessary to aid the operator in the mitigation of a beyond design basis external event. The new critical instrumentation buses perform the same function as existing instrumentation buses except they will have the added capability of obtaining primary power from DC [direct current] through their inverters connected to the station service DC power supplies.

The new equipment (inverters and critical instrumentation bus) will be installed as safety related, seismically and environmentally qualified equipment, with the primary power coming from the safety related DC station service buses, and alternate power available from the safety related AC [alternating current] essential cabinets. Therefore, the instruments being moved to the critical instrumentation bus will have a highly reliable source of power. Consequently, should the operator require the use of one of these instruments to aid in mitigating the consequences of a previously analyzed design basis event, it is highly likely that they will be available to him/her. It is therefore unlikely that the consequences of a previously evaluated accident would increase due to an inability to monitor a key containment parameter.

The TSs are being revised to add these instrument buses to the LCO [limiting condition for operation] requirements for the electrical distribution buses. No other TS LCOs are changing, no Surveillance Requirements are changing, and no instrument setpoints are changing. In fact, this TS change does not reduce any requirements. All of the components required to be Operable by the TSs before this revision request, will be required to be Operable following this change, as well as the new critical instrumentation bus. The TS requirements will therefore remain the same for the instruments being powered from the new critical instrumentation bus as well as for the instruments remaining on the AC instrument buses. In other words, the power supplies for these instruments will still be included in the TS as LCO requirements, as they were before the design change to add the critical instrumentation buses. The TS requirements will therefore continue to ensure that these indicators remain Operable during design basis events.

For the above reasons, revising the TS to include the new critical instrument buses in the electrical bus distribution Limiting Condition for Operation does not increase the probability, or consequences, of a previously analyzed event.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

TS LCO 3.8.7 is being changed to add the new critical instrumentation bus. No new modes of operation or new failure modes result from the actual TS change to any system intended for the prevention of accidents.

The design function of the instruments being moved from the existing instrument buses to the critical instrumentation buses will not change. Also, the operation of these instruments during any type of event is not changing. Only their power supply is being changed and thus no new modes of operation are created for these instruments. It is true that new components are being introduced, i.e., the inverters and instrumentation buses, thus introducing a potential failure that would not be present before the modification. However, their failure cannot cause a new or different type of accident. Furthermore the addition of these instruments will not affect any other system intended for the prevention of accidents.

The design change does not impact the existing essential cabinets or instrument buses, except to remove some loads from the instrument bus. Consequently, the design function, operation, maintenance, and testing of these existing power supplies will not change.

Finally, the new inverters and the critical instrumentation buses are not potential accident initiators; they are not intended to prevent an accident in that they do not serve as a barrier to the release of radiation either from the direct fission product boundary, or from the containment. Rather, they are intended to power instruments which serve the operators in their attempt to mitigate the consequences of accidents. Therefore, failure of these power supplies, or failure of any instrument being powered from them, cannot create an accident.

For the above reasons, the proposed amendment will not create the possibility of a new or different type of accident.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The new critical instrumentation buses being referenced in the TS will power several instruments currently being powered by the safety related instrument bus. The new inverters and critical instrumentation buses will also be safety related, as will their primary power source, the DC station service buses. Additionally, the inverters are alternately powered from the safety related essential cabinets. Therefore, because of the reliability and diversity of power supplies, the margin of safety of a loss of power event to the relocated instruments is not significantly reduced.

Loading calculations confirm that adequate design margin still exists for the DC station service buses with respect to their loading for design basis events, even with the additional loads of the added instruments.

Additionally, area heat load calculations were performed for the 130 foot elevation of the Units 1 and 2 Control Buildings which account for the new inverters, instrumentation bus and supporting components. These calculations concluded that there are no adverse effects on the [Final Safety Analysis Report] FSAR design functions.

Adding the critical instrumentation buses to the TS ensures that the new power supplies to the safety related instruments have the same TS requirements as their previous power supply. Therefore, no TS requirements have been eliminated or reduced.

For the above reasons, the margin of safety is not significantly reduced.

On the basis of the evaluation above provided by the licensee, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Leigh D. Perry, SVP & General Counsel of Operations and Nuclear, Southern Nuclear Operating Company, Inc., 40 Inverness Center Parkway, Birmingham, AL 35242.

NRC Branch Chief: Robert Pascarelli.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, (NAPS) Louisa County, Virginia

Date of amendment request: June 30, 2014. A publicly-available version is in ADAMS under Accession No. ML14183B318.

Description of amendment request: The proposed license amendment requests the changes to the Technical Specification (TS) TS 5.5.15, ''Containment Leakage Rate Testing Program,'' by replacing the reference to Regulatory Guide (RG) 1.163, ''Performance-Based Containment Leak-Test Program,'' with a reference to Nuclear Energy Institute (NEI) topical report NEI 94–01, Revision 3–A, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'' as the implementation document used to develop the North Anna performance-based leakage testing program in accordance with Option B of 10 CFR Part 50, Appendix J.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1—Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment involves changes to the NAPS Containment Leakage Rate Testing Program. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators.

Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94–01, Revision 3–A, for development of the NAPS performance-based testing program. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less than the values assumed in the plant safety analyses. The potential consequences of extending the ILRT [integrated leak rate test] interval to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in RG 1.174 [''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific changes to the Licensing Basis'']. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. NAPS has determined that the increase in Conditional Containment Failure Probability due to the proposed change is very small.

Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94–01, Revision 3–A, for the development of the NAPS performance-based leakage testing program, and establishes a 15-year interval for the performance of the containment ILRT. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3—Does this change involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94–01, Revision 3–A, for the development of the NAPS performance-based leakage testing program, and establishes a 15-year interval for the performance of the containment ILRT. This amendment does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the Containment Leakage Rate Testing Program, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests will be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94–01, Revision 3–A.

Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current NAPS PRA [probabilistic risk assessment] model concluded that extending the ILRT test interval from 10 years to 15 years results in a small change to the NAPS risk profile.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar Street, RS–2, Richmond, VA 23219.

NRC Branch Chief: Robert Pascarelli.

III. Notice of Issuance of Amendments to Facility Operating Licenses and Combined Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

A notice of consideration of issuance of amendment to facility operating license or combined license, as applicable, proposed no significant hazards consideration determination, and opportunity for a hearing in connection with these actions, was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.22(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items can be accessed as described in the ''Obtaining Information and Submitting Comments'' section of this document.

Duke Energy Carolinas, LLC, Docket Nos. 50–269, 50–270 and 50–287, Oconee Nuclear Station, Units 1, 2 and 3, Oconee County, South Carolina

Date of application for amendments: December 16, 2011, as supplemented by letters dated January 20, March 1, March 16, April 18, July 11, July 20, August 31, and November 2, 2012; April 5, June 28, August 7, and December 18, 2013; and February 14, April 3, April 11, and July 24, 2014.

Brief description of amendments: The amendments revised the Technical Specifications and the Updated Final Safety Analysis Report to add the new Protected Service Water (PSW) System to the plant's licensing basis as an additional method of achieving and maintaining safe shutdown of the reactors in the event of a high-energy line break or a fire in the turbine building, which is shared by all three units.

Date of Issuance: August 13, 2014.

Effective date: As of the date of issuance and shall be implemented within 30 days from the date of issuance.

Amendment Nos.: 386, 388, and 387. A publicly-available version is in ADAMS under Accession No. ML14206A790. Documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the license and the TSs.

*Date of initial notice in **Federal Register:*** July 10, 2012, (77 FR 40652). The supplemental letters dated January 20, March 1, March 16, April 18, July 11, July 20, August 31, and November 2, 2012; April 5, June 28, August 7, and December 18, 2013; and February 14, April 3, April 11, and July 24, 2014, provided additional information that clarified the application, did not expand the scope of the application as noticed, and did not change the staff's proposed no significant hazards consideration determination as published in the **Federal Register.**

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 13, 2014.

No significant hazards consideration comments received: No.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit 1, Washington County, Nebraska

Date of amendment request: August 5, 2013, as supplemented by letter dated January 28, 2014.

Brief description of amendment: The amendment revised the structural design basis related to the leak-before-break analysis for the reactor coolant system piping described in Section 4.3.6 of the Updated Safety Analysis Report.

Date of issuance: August 7, 2014.

Effective date: As of the date of issuance and shall be implemented within 90 days from the date of issuance.

Amendment No.: 276. A publicly-available version is in ADAMS under Accession No. ML14209A027; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR–40: The amendment revised the design basis as described in the Updated Safety Analysis Report.

*Date of initial notice in **Federal Register:*** April 8, 2014 (79 FR 19400).

The Commission's related evaluation of the amendment is contained in a safety evaluation dated August 7, 2014.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of application for amendment: August 20, 2012, as supplemented by letters dated October 25, 2012, November 8, 2012, July 2, 2013, and June 16, 2014.

Brief description of amendments: The amendments revise the condensate storage tank level requirement specified in Technical Specification surveillance requirement 3.7.6.1.

Date of issuance: August 15, 2014.

Effective date: As of its date of issuance and shall be implemented within 60 days from the date of issuance.

Amendment Nos.: Unit 1—195, Unit 2—191. A publicly-available version is in ADAMS under Accession No. ML14155A302; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Operating License Nos. NPF–2 and NPF–8: The amendments revised the Renewed Facility Operating Licenses and Technical Specifications.

*Date of initial notice in **Federal Register:*** January 15, 2013 (78 FR 3037). The supplemental letters dated October 25, 2012, November 8, 2012, July 2, 2013, and June 16, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register.**

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 15, 2014.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia and Docket No. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendment: June 26, 2013, as supplemented by letter dated January 23, 2014.

Brief description of amendment: The license amendments approve the generic application of Appendix D, ''Qualification of the ABB–NV and WLOP Critical Heat Flux (CHF) Correlations in the Dominion VIPRE–D Computer Code,'' to Fleet Report DOM–NAF–2–A, ''Reactor Core Thermal-Hydraulics Using the VIPRE–D Computer Code,'' the plant-specific applications of Appendix D to Fleet Report DOM–NAF–2–A to North Anna and Surry Power Stations, an added Surry reactor core safety limit, an increase in the Surry Minimum Temperature for Criticality (MTC), and modified references to MTC.

Date of issuance: August 12, 2014.

Effective date: As of the date of issuance and shall be implemented within 60 days from the date of issuance.

Amendment Nos.: 271, 253, 283, and 283. A publicly-available version is in ADAMS under Accession No. ML14169A359. Documents related to these amendments are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License Nos. NPF–4 and NPF–7, DPR–32 and DPR–37: Amendments changed the licenses.

*Date of initial notice in **Federal Register:*** September 3, 2013 (78 FR 54292). The supplemental dated January 23, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staffs original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 12, 2014.

No significant hazards consideration comments received: No.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: September 23, 2013.

Brief description of amendment: The amendment revised Technical Specification (TS) 5.6.5, ''CORE OPERATING LIMITS REPORT (COLR),'' to replace the methodology of Westinghouse Electric Company LLC topical report WCAP–11596–P–A, ''Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,'' with WCAP–16045–P–A, ''Qualification of the Two-Dimensional Transport Code PARAGON,'' and WCAP–16045–P–A, Addendum 1–A, ''Qualification of the NEXUS Nuclear Data Methodology,'' to determine core operating limits.

Date of issuance: August 7, 2014.

Effective date: As of its date of issuance and shall be implemented prior to core reload during Refueling Outage 20, currently expected to begin in January 2015.

Amendment No.: 209. A publicly-available version is in ADAMS under Accession No. ML14156A246; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. NPF–42. The amendment revised the Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** December 10, 2013 (78 FR 74186).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 2014.

No significant hazards consideration comments received: No.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: January 23, 2014. A redacted version was provided by letter dated March 31, 2014.

Brief description of amendment: The amendment revised the Cyber Security Plan Implementation Milestone No. 8 completion date and the physical protection license condition.

Date of issuance: August 14, 2014.

Effective date: As of its date of issuance and shall be implemented within 90 days.

Amendment No.: 210. A publicly-available version is in ADAMS under Accession No. ML14209A023; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. NPF–42. The amendment revised the Operating License.

Date of initial notice in ***Federal Register:*** June 6, 2014 (79 FR 32765).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 14, 2014.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 22nd day of August 2014.

For the Nuclear Regulatory Commission.

A. Louise Lund,

Acting Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. 2014–20671 Filed 8–29–14; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (Acrs); Meeting of the ACRS Subcommittee on AP1000; Notice of Meeting

The ACRS Subcommittee on AP1000 will hold a meeting on September 17, 2014, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance with the exception of a portion that may be closed to protect proprietary information pursuant to 5 U.S.C. 552b(c)(4). The agenda for the subject meeting shall be as follows:

Wednesday, September 17, 2014—8:30 a.m. until 12:00 p.m.

The Subcommittee will review a design change concerning the condensate return to the In-Containment Refueling Water Storage Tank. The Subcommittee will hear presentations by and hold discussions with the NRC staff, Westinghouse, and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the Full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official (DFO), Mr. Peter Wen (Telephone 301–415–2832 or Email: *Peter.Wen@nrc.gov*) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Thirty-five hard copies of each presentation or handout should be provided to the DFO thirty minutes before the meeting. In addition, one electronic copy of each presentation should be emailed to the DFO one day before the meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the DFO with a CD containing each presentation at least thirty minutes before the meeting. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on November 8, 2013, (78 FR 67205–67206).

Detailed meeting agendas and meeting transcripts are available on the NRC Web site at *http://www.nrc.gov/reading-rm/doc-collections/acrs*. Information regarding topics to be discussed, changes to the agenda, whether the meeting has been canceled or rescheduled, and the time allotted to present oral statements can be obtained from the Web site cited above or by contacting the identified DFO. Moreover, in view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with these references if such rescheduling would result in a major inconvenience.

If attending this meeting, please enter through the One White Flint North building, 11555 Rockville Pike, Rockville, MD. After registering with security, please contact Mr. Theron Brown (Telephone 240–888–9835) to be escorted to the meeting room.

Dated: August 19, 2014.

Cayetano Santos,

Chief, Technical Support Branch, Advisory Committee on Reactor Safeguards.

[FR Doc. 2014–20815 Filed 8–29–14; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[NRC–2014–0001]

Sunshine Act Meeting Notice

DATE: Weeks of September 1, 8, 15, 22, 29, October 6, 13, 2014.